NON-PUBLIC?: N
ACCESSION #: 8906270186
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Sequoyah Nuclear Power Plant Unit 1 PAGE: 1 of 4

DOCKET NUMBER: 05000327

TITLE: Reactor trip signal resulting from the closure of the main feedwater regulator valves on loss of power to the valve controllers due to personnel error.
EVENT DATE: 02/10/89 LER #: 89-005-01 REPORT DATE: 06/15/89

OPERATING MODE: 1 POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
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Engineer

COMPONENT FAILURE DESCRIPTION:
CAUSE: SYSTEM: COMPONENT: MANUFACTURER:
REPORTABLE TO NPRDS:

SUPPLEMENTAL REPORT EXPECTED: No

ABSTRACT:

On February 10, 1989 with unit 1 in mode 1, a reactor trip occurred at 2036 EST. The trip signal was a result of a steam generator (SG) steam flow to feedwater flow mismatch of greater than 40 percent of the nominal value of steam flow at full power coincident with a low SG level (25 percent) signal on SG loop 3. Two Instrument Maintenance (IM) technicians were implementing work request (WR)-B238429 on flow recorder (FR)-2-200/201, "Condenser Bypass/Makeup Flow," which is located on main control room (MCR) panel 1-M-3. The recorder pen needed to be restrung which required it to be removed from the case. The technicians fully removed the recorder in its case which required lifting the power supply leads. After reinstallation, the technicians reterminated the power supply leads, at which time one technician determined the terminating leads were too close to each other. The technician inadvertently shorted a screwdriver between the terminals, tripping open breaker No. 39 on 120-VAC vital instrument board I-II which is the power supply to a plug mold supplying the recorder. The plug mold is also the common power supply to the flow

indicating controllers (FIC)-3-35, -90, and -103 which control main feedwater regulating valves (MFWRVs) FCV-3-35, -90, and -103 for SG loops 1, 3, and 4. FIC-3-48 for MFWRV on SG loop 2 is powered from board I-III. The three MFWRVs closed, isolating main feedwater (MFW) to the SGs. This resulted in the solid state protection system reactor trip signal. The root cause of the reactor trip signal was personnel error, in that, appropriate precautions were not taken in performing terminations of energized equipment. To prevent recurrence of this event, this event was discussed with IM planners, technicians, and engineers to familiarize them with the root cause of the event. Additionally, open WRs and scheduled preventive maintenances on MCR recorders were retrieved by the Work Control Group for replanning to address precautions on equipment common power supplies. In addition to the above, SQM2, "Maintenance Management System," and SQM2.2, "Maintenance Management System Troubleshooting," will be revised to clarify the requirements for working on energized equipment.

END OF ABSTRACT

DESCRIPTION OF EVENT

On February 10, 1989 with unit 1 in mode 1 (100-percent reactor power, 2230 psig, 578 degrees F), a reactor trip occurred at 2036 EST. The trip signal was a result of a steam flow to feedwater flow mismatch of greater than 40 percent of the nominal value of steam flow at full power coincident with a low steam generator (SG) (EIIS Code AB) level (25 percent) signal being present on SG loop 3. Before the trip, work was in progress on work request (WR)-B238429 to repair and recalibrate flow recorder (FR)-2-200/201, "Condenser Bypass/Makeup Flow."

On February 9, 1989, two Instrument Maintenance (IM) technicians were assigned to work WR-B238429 on FR-2-200/201, which is located on main control room (MCR) (EIIS Code NA) panel 1-M-3. The recorder pen needed to be restrung which required it to be removed. Normally, removal is performed by sliding the recorder out of its case and disconnecting the power supply leads on the recorder, however, when this was attempted the recorder became lodged in the case. Subsequently, the technicians fully removed the recorder in its case which required lifting the power supply leads. The recorder work was completed on February 9, 1989, and the recorder was reinstalled. On February 10, 1989, the power supply leads were reterminated, at which time one technician determined the terminating leads were too close to each other. The technician inadvertently shorted a screwdriver between the terminals when attempting to realign the connections. The short caused breaker No. 39 on 120-VAC vital instrument board I-II (EIIS Code EE) to trip open which powers the plug mold used by the recorder.

The plug mold is also the common power supply to the flow indicating controllers (FIC)-3-35, -90, and -103 which control main feedwater regulating valves (MFWRVS) FCV-3-35, -90, and -103 for SG loops 1, 3, and 4, respectively. FIC-3-48 for the MFWRV for SG loop 2 is powered from 120-VAC vital instrument board I-III. The loss of power to the FICs caused the three MFWRVs to close, thereby isolating main feedwater to the SGs. This resulted in a steam flow to feedwater flow mismatch signal on loops 1, 3, and 4 and the generation of the solid state protection system (SSPS) reactor trip signal at 2036 EST when the level in SG loop 3 decreased to 25 percent which co pleted
the required SSPS logic of a flow mismatch and low level in any one SG.

After the trip, Operations personnel responded using emergency procedures E-0, "Reactor Trip or Safety Injection, ES-0.1, "Reactor Trip Response," and General Operating Instruction (GOI)-3, "Plant Shutdown From Minimum Load To Cold Shutdown," to stabilize the unit. A notification of unusual event (NOUE) in accordance with Emergency Plan Implementing Procedure (EPIP)-2, "Notification of Unusual Event," was declared at 2100 EST due to possible steam line leakage. It was later verified that no steam line leak existed, and the NOUE was exited at 2158 EST. The plant steam dump control system (SDCS) (EIIS Code SE) operated as expected. The auxiliary feedwater (AFW) (EIIS Code BA) pumps started on a MFW isolation which occurred as designed after the trip when Tavg decreased to 554 degrees F and began injecting greater than 440 gpm to each SG. Manual control of AFW was taken when Tavg was less than 547 degrees F in accordance with ES-0.1. Under manual control, the turbine-driven AFW pump was ramped to minimum speed, and the motor-driven AFW valves were placed in manual bypass to control flow to each SG at 110 gpm. Loop 3 and 4 AFW valves performed as expected when placed in manual bypass, however, loop 1 and 2 valves did not respond as the balance of plant operator expected, and the valves were manually ramped closed at the MCR panel.

DESCRIPTION OF EVENT (Continued)

When Tavg was less than 540 degrees F, the operator commenced emergency boration at a rate of 80 gpm in accordance with ES-0.1. This was continued for 9 minutes for a total boration delivery of 720 gallons. The reactor coolant system (RCS) (EIIS Code AB) Tavg decreased to approximately 537 degrees F. The plant was brought to a stabilized condition with no adverse effect on the plant or public safety. During recovery from the trip, the turbine overspeed first-out annunciator alarmed for no apparent reason.

Subsequent to the trip, the IM technicians heard MCR alarms and notified the back shift IM supervisor. The shift operations supervisor (SOS) questioned the technicians on the work being performed and requested they check the supply breaker for the plug mold. Breaker No. 39 on 120-VAC vital instrument board I-II was found tripped. The breaker was reset and closed by Operations, and power was returned to the instrumentation.

At 0039 EST on February 11, 1989, steam dump valve FCV-1-103 operation became erratic and subsequently popped open which caused RCS temperature to decrease to 520 degrees F. To stop the ensuing cooldown, the operators closed the main steam isolation valves (MSIVs) and locally isolated the valve. At 0230 EST, it was determined that an NOUE should have been entered at 0039 EST and exited when the MSIVs were closed. At 0239 EST, the NRC was notified of this event in accordance with 10 CFR 50.72, paragraph a.1.i.

CAUSE OF EVENT

The root cause of the generation of the reactor trip signal was personnel error, in that, appropriate precautions were not taken in performing terminations of energized equipment. The reactor trip resulted from a steam flow to feedwater flow mismatch coincident with a low SG level on loop 3 when the MFWRVs were closed. The closing of the valves was a result of loss of power to the valve controllers when the technicians were attempting to terminate the power leads to recorder FR-2-200/201. During the termination process, the technicians inadvertently shorted the recorder power supply. This tripped the breaker feeding the plug mold which included the control power for the MFWRV controllers. The cause of the turbine overspeed annunciation during trip recovery has been determined to be an induced voltage signal affecting the overspeed circuitry from an unknown source. The cause of FCV-1-103 erratic operation has been determined to be a disconnected positioner feedback arm.

ANALYSIS OF EVENT

This event is being reported in accordance with 10 CFR 50.73, paragraph a.2.iv., as an event that resulted in an automatic actuation signal of the reactor protection system (RPS). Notification to the NRC was made in accordance with 10 CFR 50.72, paragraph a.1.i at 2120 EST on February 10, 1989.

This reactor trip signal (steam flow to feedwater flow mismatch coincident with low SG level) is required operable while in modes 1 and 2 and is present as an anticipation of a loss of secondary heat sink. The reactor trip signal was a result of maintenance activities resulting in an actual secondary-side mismatch condition. The safety-related RPS logic actuated as designed to mitigate the consequences of an anticipated loss of secondary heat sink event. In this event, there was no actual loss of secondary heat sink; therefore, this event did not impact the health and safety of the general public.

CORRECTIVE ACTIONS

Immediate corrective actions were to initiate reactor trip recovery using E-0 and ES-0.1. An immediate investigation was also initiated into the cause of the reactor trip signal and to determine any corrective actions necessary before returning the unit to operation.

To prevent recurrence of this event, this event was discussed with IM planners, technicians, and engineers to familiarize them with the root cause of the event. Additionally, open WRs and scheduled preventive maintenances on MCR recorders were retrieved by the Work Control Group for replanning to address precautions on equipment common power supplies. Information on generic problems associated with multiple power feeds for instrumentation will be incorporated into IM initial training through the Industry/Sequoyah Experience Report familiarization lesson plan by September 1, 1989.

In addition to the above, SQM2, "Maintenance Management System," and SQM2.2 "Maintenance Management System Troubleshooting," will be revised by June 30, 1989, to clarify the requirements for working on energized equipment. This will require that equipment be deenergized if possible; and, when in the judgment of planners this is not feasible, precautions will be listed stating the potential effects that could be experienced in the event of personnel error or mistake.

No problem was detected with the operation of the turbine overspeed circuitry during troubleshooting under WR-B797075 on February 12, 1989. No problem was detected with the operation of the loops 1 and 2 AFW level control valves during troubleshooting unders WR-B797072 and WR-B797943. The positioner feedback arm on FCV-1-103 was discovered ' disconnected during performance of WR-B797963. The arm was reconnected and the valve stroke was verified by latest calibration documentation.

ADDITIONAL INFORMATION

There have been four previous reported instances of a reactor trip from a steam flow to feedwater flow mismatch coincident with a low SG level: SQRO-50-327/84054, -327/88036, -328/88023, and -328/88027.

COMMITMENTS

1. Information on generic problems associated with multiple power feeds for instrumentation will be incorporated into IM initial training through the Industry/Sequoyah Experience Report familiarization lesson plan by

September 1, 1989.

2. SQM and SQM2.2 will be revised by June 30, 1989, to clarify the requirements for working on energized equipment. This will require that equipment be deenergized if possible; and, when in the judgment of planners this is not feasible, precautions will be listed stating the potential effects that could be experienced in the event of personnel error or mistake.

ATTACHMENT 1 TO 8906270186 PAGE 1 OF 1

TENNESSEE VALLEY AUTHORITY

6N 38A Lookout Place

June 16, 1989

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Gentlemen:

TENNESSEE VALLEY AUTHORITY - SEQUOYAH NUCLEAR PLANT UNIT 1 - DOCKET NO.
50-327 - FACILITY OPERATING LICENSE DPR-77 - LICENSEE EVENT REPORT (LER)
327/89005, REVISION 1

The enclosed revised LER provides additional information concerning a unit 1 reactor trip signal generated from a steam generator steam flow to feedwater flow mismatch coincident with a low steam generator level signal. This condition resulted from the closure of the main feedwater regulating valves on loss of power to the valve controllers. This event was reported in accordance with 10 CFR 50.73, paragraph a.2.iv, on March 7, 1989.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

J. R. Bynum, Vice President
Nuclear Power Production

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